ATTACHMENT 1

RADIOLOGICAL ACCIDENT ANALYSIS for DECOMMISSIONING WARD CENTER for NUCLEAR STUDIES at CORNELL UNIVERSITY

FACILITIES INVENTORY BLDG. NO. 2061

Approvals Page

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1.0 PURPOSE AND OBJECTIVES

This calculation provides an analysis of the potential radiological accidents that could occur during decommissioning of the Cornell University Ward Center and affect the public or occupational health and safety. Prior to the initiation of decommissioning activities all reactor fuel will be removed from the Ward Center. Therefore potential accidents involving reactor fuel were not considered. The accident analyses show that the doses to the public from potential accidents are below the U.S. EPA Protective Action Guides (PAGs) that have been developed to protect members of the public from the consequences of accidents (EPA 400-R-92-001, 1992). Therefore, no new protective measures are required to protect public or occupational health and safety. Bounding analyses of potential accidents at a level of detail consistent with existing information about the radiological hazards at Ward Center were performed.

2.0 CONCLUSIONS

The accident analysis shows that the postulated accident scenarios would result in TEDEs to a member of the public at the site boundary that are much less than the U.S. EPA's lower PAG of 1 rem (1000 mrem) (USEPA 1992) and the NRC dose limits for individual members of the public of 0.1 rem/yr (100 mrem/yr) (10 CFR 20.1301).

The results of the accident analysis show that off-site consequences from accidents are well below the U.S. EPA's PAGs and the NRC's dose limits for individual members of the public; therefore, off-site emergency plans are not needed.

3.0 ASSUMPTIONS/INPUTS

3.1 Assumptions

The following assumptions were used in the accident analyses:

- The radionuclide inventories were based on the data provided in the NUREG/CR-1756 (Ref. 4.4) and *Characterization Survey Report for Ward Center for Nuclear Studies at Cornell University, Facilities Inventory Bldg. No. 2061*, February 2003.
- To be conservative, unfavorable weather conditions for atmospheric dispersion were assumed. For the purposes of this analysis, atmospheric stability class F with a wind speed of 2 m/s (6.6 ft/s) was assumed, which represents a situation with minimal dispersion of a potential radioactive plume. In addition, the radioactive material was assumed to be released at ground level and to remain airborne as it travels downwind.
- A screening analysis approach was used for Ward Center accident analysis because the radioactive inventories are very small compared to those in operating reactors and in fuel cycle facilities subject to NRC regulation.
- The screening analysis for Ward Center consists of identifying and analyzing plausible accident scenarios that could occur during decommissioning activities.

3.2 Potential Radiological Accidents

Identifying potential accident scenarios included evaluating Ward Center areas that contain the highest inventories of radioactive material, describing energy sources and external events, reviewing proposed activities, and considering combinations of these elements that could lead to a release of radioactive material. Because of the limited inventory, the evaluation of accident scenarios conservatively assumed that no design or procedural controls would be available to prevent or mitigate accidental releases, even though such controls will be implemented during decommissioning activities. This assumption allows for a worst-case accident analysis to be performed.

3.3 Highest Radionuclide Inventories at the Ward Center

One area with a high radiological material inventory at Ward Center is the pool of the TRIGA reactor. Most of the activity in the pool is contained in activated components that include control rods, instrument thimbles, etc. During decommissioning, these components will be cut mechanically underwater and lifted and placed into transport liners underwater. A waste shipping liner containing activated hardware was estimated to contain 1,460 curies (NUREG/CR-1756-v1, Tables E1-2, E.1-3, E.1-5 and E.1-6, for Reference Research Reactor). Most of the activity is Co-60 (84%), with Fe-55 (8.6%), Mn-54 (4.7%), and smaller inventories of other radionuclides. These are simple operations, and the worst-case accident scenario would be dropping one of these filled transport liners as it is being lifted.

The demineralizer column for the TRIGA has a concentrated radionuclide inventory. During the decommissioning the resin column will be removed and placed into a transport container. The accident scenario is for the resin column to be dropped and broken open during handling. Most of the inventory is Co-60 (52%), with smaller inventories of other radionuclides, Mn-54 (19%), Zn-65 (12%), C-14 (5%), Co-58 (5%) and Ni-63 (4%).

As part of the decommissioning process there will be radioactive waste staged prior to loading it into shipping containers for transportation off-site. The accident scenario is for a fire in the dry solid waste for shipping (i.e., rags, wipes and anticontamination clothing). It was assumed that the activity levels in this material were the same as for the resin column. This is very conservative as the resin column concentrated activity and most of the dry solid waste will have minimal contamination. The calculated inventory is Co-60 (52%), with smaller inventories of other radionuclides, Mn-54 (19%), Zn-65 (12%), C-14 (5%), Co-58 (5%) and Ni-63 (4%).

Another accident scenario considered is for a fire to occur in the graphite material related to the thermal column during its removal from the reactor. The calculated inventory is C-14 (55%), H-3 (37%), Co-60 (3%), Ni-63 (2%) and Th-230 (2%).

All of the other rooms at Ward Center have smaller radioactive inventories. Therefore, the results of accident analyses conducted for decommissioning the TRIGA reactor pool and the Reactor Complex bound the potential impacts of inside accidents during decommissioning of the Ward Center.

Therefore, accident analyses were performed for these items because they bound the impacts of potential radiological accidents during the decommissioning of the Ward Center.

4.0 REFERENCES

4.1 U.S. EPA, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents*, 400-R-92-001, 1992.

4.2 10 CFR Part 20.1301, *Dose Limits for Individual Members of the Public*

4.3 10 CFR Part 20, *Standards for Protection Against Radiation*, Federal Register, Vol. 56, No. 88, NRC May 21, 1991

4.4 Duratek, Inc. *2003. Characterization Survey Report for Ward Center.*

4.5 *Technology, Safety and Costs of Decommissioning Reference Nuclear Research and Test Reactors*, NUREG/CR-1756, March 1982

4.6 International Commission on Radiological Protection (ICRP), *Limits for Intakes of Radionuclides by Workers*, 1979, ICRP Publication No. 30, Annals of the ICRP Vol. 2, No. 3/4

4.7 *Federal Guidance Report No. 11*, EPA-520/1-88-020, September 1988

4.8 *Federal Guidance Report No. 12*, EPA-402-R-93-081, September 1993.

5.0 POTENTIAL ACCIDENT SCENARIO'S

Considering the planned decommissioning activities, accident scenarios that could result in releasing radioactive material as airborne particles small enough to be respirable were evaluated. Such releases could occur during a fire in a waste storage area, as a result of dropping a resin column or as a result of dropping of a container of activated hardware. Because the Ward Center is outside of the 100-year floodplain and releases from soil areas near the river will be minimal based on historical erosion information, extreme precipitation events are not expected to cause off-site radiological impacts.

Based on the decommissioning activities outlined in the Ward Center Decommissioning Plan and the radiological inventories identified in Section 3.3, the following accident scenarios were evaluated:

- A waste shipping liner containing activated hardware could be dropped while moving it from the pool to a transportation cask.
- A demineralizer resin column could also be dropped and then burst open and release airborne particles.
- The potential for fires was also considered. A fire in an area used stage dry solid waste for shipping (i.e., rags, wipes and anticontamination clothing).
- A fire in a block of activated graphite material was also considered.

6.0 METHODOLOGY FOR CALCULATING TOTAL EFFECTIVE DOSE EQUIVALENT

The consequences of accidents were quantified by calculating the TEDE to a member of the public at the site boundary. Then the calculated TEDE was compared to the U.S. EPA's lower PAG of 1 rem (1,000 mrem) and the NRC 0.1 rem/yr (100 mrem) dose limit for a member of the public, to determine whether or not the calculated exposure is acceptable. Equation 6-1 was used to calculate the TEDE[1]:

$$TEDE_i = CEDE_i + Ext_i \qquad (6\text{-}1)$$

Where:

TEDE	=	total effective dose equivalent
CEDE	=	committed effective dose equivalent
Ext	=	contribution from external irradiation
i	=	radionuclide.

The committed effective dose equivalent (CEDE) is the dose contribution from inhalation as the cloud passes by the receptor. Consistent with the lung model developed by the International Commission on Radiological Protection (ICRP 1979), the CEDE is found by:

$$CEDEi = Q_i\,(x/Q) \times B \times D_i \times t \qquad (6\text{-}2)$$

Where:

Q_i	=	the released activity of nuclide *I* per emit time, in Ci/sec
t	=	the duration of the release, in sec. The product of $Q_i$xt is equal to the activity of nuclide *i* released. For the accident scenario's in this document there is a rapid release of activity, which is assumed to occur in one second.
x/Q	=	the airborne dosage (concentration integrated over the duration of cloud passage) per unit activity released, in s/m^3. The derivation of x/Q presented in Appendix A shows that for a distance of 100 meters (0.06 mi) in atmospheric stability class F with a wind speed of 2 m/s, x/Q = 4.52×10^{-3} s/m^3.
B	=	the breathing rate, typically 3.3×10^{-4} m^3/s. (This is the breathing rate for adults during light activity [ICRP 1979]).
D_i	=	the factor that converts the amount of activity inhaled into the CEDE. Values of D*i* are given in Federal Guidance Report No. 11 (EPA-520/1-88-020, September 1988).

The dose contribution from external irradiation is found by:

$$Ext_i = Q_i \times (x/Q) \times F_i \times t \qquad (6\text{-}3)$$

Where: F_i = the dose coefficient for air submersion. Values of F_i are given in Federal Guidance Report No. 12 (EPA-402-R-93-081, September 1993).

[1] This estimate of the TEDE neglects any contribution from gamma rays emitted by radionuclides deposited on the ground. Such doses build up relatively slowly and, if necessary, can be controlled by various countermeasures.

7.0 ANALYSIS OF POTENTIAL ACCIDENT SCENARIO'S

7.1 Scenario 1: Release While Moving Irradiated Hardware Liner

Most of the activity in the TRIGA pool is contained in activated components that include control rods, instrument thimbles, etc. Because cutting operations for components will be performed underwater, no cutting accident releases were postulated. However a liner filled with irradiated hardware could be dropped while it is lifted for placement into a shipping container. The activity in irradiated hardware is contained within the metal structure of the hardware item except for surface contamination. It would be highly unlikely for a component in the liner to break. If it did break, the diameters of any particles produced would be large enough that it is unlikely that the particles would remain airborne and be respirable. However, even though it is not plausible that an accident could result in measurable exposures at the site boundary; this scenario was evaluated because it includes the largest curie inventory and it demonstrates that potential exposures to the public are acceptable even when worst case assumptions are utilized.

A waste shipping liner containing 120 cubic feet of activated hardware was estimated to contain 1,460 curies (NUREG/CR-1756-v1, Tables E1-2, E.1-3, E.1-5 and E.1-6, for Reference Research Reactor). Most of the activity is Co-60 (84%), with Fe-55 (8.6%), Mn-54 (4.7%), and smaller inventories of other radionuclides. The worst-case accident scenario would be dropping the filled liner as it is being lifted. If 1 percent of the activity of this liner was respirable and 1 percent of the respirable material escaped the liner and became airborne during the accident (i.e., approximately 5.4×10^2 g, assuming a waste density of 1.6 g/cm^3), the airborne quantities of radionuclides would be 1.22×10^{-1} Ci of Co-60, 1.25×10^{-2} Ci of Fe-55, 6.83×10^{-3} Ci of Mn-54, and lesser quantities of other radionuclides. Using the values of χ/Q and B given in Section 6.0 in Equations 6-1 through 6-3, the CEDE, the contribution from external irradiation (Ext), and the TEDE were calculated as shown in Table 7-1.

Table 7-1. TEDE Calculation Table for Scenario 1: Hardware Liner Drop

Nuclide *i*	*Lung Clearance Class**	Q_i (Ci)	D_i (mrem/Ci)	F_i ([mrem/s]/[Ci/m^3])	CEDE$_i$ (mrem)	Ext$_i$ (mrem)	TEDE$_i$ (mrem)
C-14	Special	3.90E-06	2.09E+06	8.30E-04	1.21E-05	1.46E-11	1.21E-05
Mn-54	W	6.83E-03	6.70E+06	1.51E+02	6.83E-02	4.68E-03	7.29E-02
Fe-55	W	1.25E-02	1.34E+06	0.00E+00	2.48E-02	0.00E+00	2.48E-02
Co-60	Y	1.22E-01	2.19E+08	4.67E+02	3.97E+01	2.57E-01	4.00E+01
Ni-59	W	2.36E-05	9.18E+05	0.00E+00	3.24E-05	0.00E+00	3.24E-05
Ni-63	W	2.71E-03	2.30E+06	0.00E+00	9.29E-03	0.00E+00	9.29E-03
Zn-65	Y	1.82E-03	2.04E+07	1.03E+02	5.55E-02	8.52E-04	5.63E-02
Nb-93m	W	4.31E-09	2.95E+07	1.64E-02	1.90E-07	3.21E-13	1.90E-07
Nb-94	W	5.58E-08	4.41E+08	2.85E+02	3.45E-05	7.20E-08	3.46E-05
Total		**1.46E-01**	--	--	**3.99E+01**	**2.62E-01**	**4.01E+01**

* C-14 Special Models, Mn-54 oxides correspond to lung clearance Class W, Fe-55 oxides Class W, Nickel oxides Class W, and Zn-65 all compounds Class Y, Niobium oxides class W, and Co-60 oxides Class Y, from Federal Guidance Report # 11 (USEPA 1988), Table 3.

As shown in Table 7-1, the TEDE is less than 41 mrem, to which the external dose is a negligible contributor. The TEDE of less than 41 mrem derived using conservative conditions is less than the U.S. EPA's lower PAG of 1 rem (1,000 mrem) (USEPA 1992) and NRC dose limits for individual members of the public of 0.1 rem/yr (10 CFR 20.1301).

7.2 Scenario 2: Release from a Dropped Demineralizer Column

The TRIGA demineralizer has a concentrated radionuclide inventory. During the decommissioning, the resin column will be removed and placed into a transport container. The accident scenario is for the resin column to be dropped and broken open during handling. The activity is contained within the resin beads inside the demineralizer column except for surface contamination. If the column was dropped and broken the diameters of any particles produced would be large enough that it is unlikely that the particles would remain airborne and be respirable. However, even though it is not plausible that an accident could result in measurable exposures at the site boundary, this scenario was evaluated because it is one of the few areas with a significant curie inventory and it demonstrates that potential exposures to the public are acceptable even when worst case assumptions are utilized.

A demineralizer resin column containing 10.8 cubic feet of resin was estimated to contain 0.2 millicuries based upon an independent laboratory analysis of the resin. Most of the inventory is Co-60 (52%), with smaller inventories of other radionuclides, Mn-54 (19%), Zn-65 (12%), C-14 (5%), Co-58 (5%) and Ni-63 (4%). The worst-case accident scenario would be dropping the resin column liner as it is being lifted. If 1 percent of the activity of this column was respirable and 10 percent of the respirable material escaped the column and became airborne during the accident (i.e., approximately 337 g, assuming a waste density of 1.1 g/cm^3), the airborne quantities of radionuclides would be 1.05×10^{-07} Ci of Co-60, 3.84×10^{-08} Ci of Mn-54, 2.42×10^{-08} Ci of Zn-65, 1.09×10^{-08} Ci of C-14, 7.15×10^{-09} Ci of Ni-63and lesser quantities of other radionuclides. Using the values of χ/Q and B given in Section 6.0 in Equations 6-1 through 6-3, the CEDE, the contribution from external irradiation (Ext), and the TEDE were calculated as shown in Table 7-2.

Table 7-2. TEDE Calculation Table for Scenario 2: Dropped Demineralizer Column

Nuclide *i*	*Lung Clearance Class**	Q_i (Ci)	D_i (mrem/Ci)	F_i ([mrem/s]/[Ci/m^3])	CEDE$_i$ (mrem)	Ext$_i$ (mrem)	TEDE$_i$ (mrem)
H-3	Special	1.60E-09	6.40E+04	1.23E-03	1.53E-10	8.87E-15	1.53E-10
C-14	Special	1.09E-08	2.09E+06	8.30E-04	3.38E-08	4.07E-14	3.38E-08
Mn-54	W	3.84E-08	6.70E+06	1.51E+02	3.84E-07	2.63E-08	4.10E-07
Co-57	Y	6.74E-10	9.07E+06	2.08E+01	9.12E-09	6.33E-11	9.18E-09
Co-58	Y	9.37E-09	1.09E+07	1.76E+02	1.52E-07	7.47E-09	1.60E-07
Co-60	Y	1.05E-07	2.19E+08	4.67E+02	3.42E-05	2.21E-07	3.44E-05
Ni-63	W	7.15E-09	2.30E+06	0.00E+00	2.45E-08	0.00E+00	2.45E-08
Zn-65	Y	2.42E-08	2.04E+07	1.03E+02	7.35E-07	1.13E-08	7.46E-07
Sb-124	W	1.88E-09	2.52E+07	3.39E+02	7.05E-08	2.88E-09	7.34E-08
Cs-134	D	1.62E-09	4.63E+07	2.80E+02	1.11E-07	2.05E-09	1.13E-07
Cs-137	D	1.39E-09	3.19E+07	2.87E-02	6.62E-08	1.80E-13	6.62E-08
Eu-152	W	1.88E-09	2.21E+08	2.09E+02	6.20E-07	1.78E-09	6.22E-07
Th-230	Y	9.07E-11	2.62E+11	6.44E-02	3.54E-05	2.64E-14	3.54E-05
Th-232	Y	2.32E-11	1.15E+12	3.23E-02	3.99E-05	3.39E-15	3.99E-05
U-233/234	Y	4.82E-11	1.32E+11	6.04E-02	9.53E-06	1.32E-14	9.53E-06
U-235	Y	4.11E-11	1.23E+11	2.67E+01	7.54E-06	4.96E-12	7.54E-06
Total		**2.04E-07**	--	--	**1.29E-04**	**2.73E-07**	**1.29E-04**

* H-3 & C-14 Special Models, Mn-54 oxides correspond to lung clearance Class W, Cobalt oxides Class Y, Nickel oxides Class W, and Zn-65 all compounds Class Y, Antimony oxides Class W, Cesium all forms Class D, Thorium oxides Class Y, and Uranium oxides Class Y, from Federal Guidance Report # 11 (USEPA 1988), Table 3.

As shown in Table 7-2, the TEDE is less than 1 mrem, to which the external dose is a negligible contributor. The TEDE of less than 1 mrem derived using conservative conditions is less than the U.S. EPA's lower PAG of 1 rem (1,000 mrem) (USEPA 1992).

7.3 Scenario 3: Release from a Waste Staging Area Fire

As part of the decommissioning process there will be radioactive waste staged prior to loading it into shipping containers for transportation off-site. The accident scenario is for a fire in the dry solid waste for shipping (i.e., rags, wipes and anticontamination clothing). It was assumed that the activity levels in this material were the same as for the resin column. This is very conservative as the resin column concentrated activity and most of the dry solid waste will have minimal contamination. A waste inventory of 360 cubic feet of dry solid waste was estimated to contain 0.3 millicuries assuming it would have the same specific activity as the resin in Scenario 2. The calculated inventory is Co-60 (52%), with smaller inventories of other radionuclides, Mn-54 (19%), Zn-65 (12%), C-14 (5%), Co-58 (5%) and Ni-63 (4%).

It is estimated that combustion of this structural material would release approximately 50% of the contamination in a respirable form. The combustion of this waste under this scenario would release 7.4×10^{-5} Ci of Co-60, 2.71×10^{-5} Ci of Mn-54, 1.71×10^{-5} Ci of Zn-65, 7.66×10^{-6} Ci of C-14, 6.61×10^{-6} Ci of Co-58, 5.04×10^{-6} Ci of Ni-63, and lesser quantities of other radionuclides. Using the values of χ/Q and B given in Section 6.0 in Equations 6-1 through 6-3, the CEDE, the contribution from external irradiation (Ext), and the TEDE were calculated as shown in Table 7-3.

Table 7-3. TEDE Calculation Table for Scenario 3: Waste Staging Area Fire

Nuclide *i*	*Lung Clearance Class*	Q_I (Ci)	D_I (mrem/Ci)	F_I ([mrem/s]/[Ci/m^3])	CEDE$_i$ (mrem)	Ext$_i$ (mrem)	TEDE$_i$ (mrem)
H-3	Special	1.13E-06	6.40E+04	1.23E-03	1.08E-07	6.26E-12	1.08E-07
C-14	Special	7.66E-06	2.09E+06	8.30E-04	2.38E-05	2.87E-11	2.38E-05
Mn-54	W	2.71E-05	6.70E+06	1.51E+02	2.71E-04	1.86E-05	2.89E-04
Co-57	Y	4.76E-07	9.07E+06	2.08E+01	6.43E-06	4.47E-08	6.48E-06
Co-58	Y	6.61E-06	1.09E+07	1.76E+02	1.07E-04	5.27E-06	1.13E-04
Co-60	Y	7.40E-05	2.19E+08	4.67E+02	2.41E-02	1.56E-04	2.43E-02
Ni-63	W	5.04E-06	2.30E+06	0.00E+00	1.73E-05	0.00E+00	1.73E-05
Zn-65	Y	1.71E-05	2.04E+07	1.03E+02	5.19E-04	7.97E-06	5.27E-04
Sb-124	W	1.33E-06	2.52E+07	3.39E+02	4.97E-05	2.03E-06	5.18E-05
Cs-134	D	1.14E-06	4.63E+07	2.80E+02	7.86E-05	1.44E-06	8.01E-05
Cs-137	D	9.80E-07	3.19E+07	2.87E-02	4.67E-05	1.27E-10	4.67E-05
Eu-152	W	1.33E-06	2.21E+08	2.09E+02	4.37E-04	1.26E-06	4.39E-04
Th-230	Y	6.40E-08	2.62E+11	6.44E-02	2.50E-02	1.86E-11	2.50E-02
Th-232	Y	1.64E-08	1.15E+12	3.23E-02	2.81E-02	2.39E-12	2.81E-02
U-233/234	Y	3.40E-08	1.32E+11	6.04E-02	6.72E-03	9.28E-12	6.72E-03
U-235	Y	2.90E-08	1.23E+11	2.67E+01	5.32E-03	3.50E-09	5.32E-03
Total		1.44E-04	--	--	9.08E-02	1.93E-04	9.10E-02

* H-3 & C-14 Special Models, Mn-54 oxides correspond to lung clearance Class W, Cobalt oxides Class Y, Nickel oxides Class W, and Zn-65 all compounds Class Y, Antimony oxides Class W, Cesium all forms Class D, Thorium oxides Class Y, and Uranium oxides Class Y, from Federal Guidance Report # 11 (USEPA 1988), Table 3.

As shown in Table 7-3, the TEDE is less than 1 mrem, to which the external dose is a negligible contributor. The TEDE of less than 1 mrem derived using conservative conditions is less than the U.S. EPA's lower PAG of 1 rem (1,000 mrem) (USEPA 1992).

7.4 Scenario 4: Release from a Graphite Block Fire

As part of the decommissioning process there will be blocks of activated graphite removed. The accident scenario is for a fire in the graphite material. There is a block of graphite just over 2-feet on a side that was assumed would catch fire. A block volume of 9.3 cubic feet of graphite was estimated to contain 2.49 x 10^{-2} mCi based upon an independent laboratory analysis of a graphite sample. The calculated inventory is C-14 (55%), H-3 (37%), Co-60 (3%), Ni-63 (2%) and Th-230 (2%) with smaller inventories of other radionuclides.

It is estimated that combustion of this structural material would release approximately 50% of the contamination in a respirable form. The combustion of this waste under this scenario would release 6.80 x 10^{-6} Ci of C-14, 4.64 x 10^{-6} Ci of H-3, 3.45 × 10^{-7} Ci of Co-60, 2.74 x 10^{-7} Ci of Th-230, 2.25 x 10^{-7} Ci of Ni-63, and lesser quantities of other radionuclides. Using the values of χ/Q and B given in Section 6.0 in Equations 6-1 through 6-3, the CEDE, the contribution from external irradiation (Ext), and the TEDE were calculated as shown in Table 7-4.

Table 7-4. TEDE Calculation Table for Scenario 4: Graphite Block Fire

Nuclide i	Q_i (Ci)	D_i (mrem/Ci)	F_i ([mrem/s]/[Ci/m^3])	CEDE$_i$ (mrem)	Ext$_i$ (mrem)	TEDE$_i$ (mrem)
H-3	4.64E-06	6.40E+04	1.23E-03	4.43E-07	2.57E-11	4.43E-07
C-14	6.80E-06	2.09E+06	8.30E-04	2.12E-05	2.55E-11	2.12E-05
Co-60	3.45E-07	2.19E+08	4.67E+02	1.13E-04	7.28E-07	1.13E-04
Ni-63	2.25E-07	2.30E+06	0.00E+00	7.71E-07	0.00E+00	7.71E-07
Th-227	3.24E-08	1.62E+12	1.81E+01	7.80E-02	2.64E-09	7.80E-04
Th-230	2.74E-07	2.62+11	6.44E-02	1.07E-01	7.97E-11	1.07E-01
Th-232	3.69E-08	1.15E+12	3.23E-02	6.33E-02	5.38E-12	6.33E-02
U-233/234	3.66E-08	1.32E+11	6.04E-02	7.24E-03	1.00E-11	7.24E-03
U-238	4.07E-08	1.18E+11	1.26E-02	7.18E-03	2.32E-12	7.18E-03
Total	9.90E-07			1.85E-01	7.30E-07	1.85E-01

* H-3 & C-14 Special Models, Cobalt oxides correspond to lung clearance Class Y, Nickel oxides Class W, Thorium oxides Class Y, and Uranium oxides Class Y, from Federal Guidance Report # 11 (USEPA 1988), Table 3.

As shown in Table 7-4, the TEDE is less than 1 mrem, to which the external dose is a negligible contributor. The TEDE of less than 1 mrem derived using conservative conditions is less than the U.S. EPA's lower PAG of 1 rem (1,000 mrem) (USEPA 1992).

8.0 CONCLUSIONS

The accident analysis shows that the postulated accident scenarios would result in TEDE's to a member of the public at the site boundary that are much less than the U.S. EPA's lower PAG of 1 rem (1,000 mrem) (USEPA 1992) and the NRC dose limits for individual members of the public of 0.1 rem/yr (100 mrem/yr) (10 CFR 20.1301).

The results of the accident analysis show that off-site consequences from accidents are well below the U.S. EPA's PAGs; therefore, off-site emergency plans are not needed.

9.0 APPENDICES

Appendix A DERIVATION OF χ/Q

APPENDIX A

DERIVATION OF χ/Q

In the accident analysis presented in Section 6.0, the quantity χ/Q is used to express the dilution of the released effluent as it travels 100 meters (0.06 mi) to the site boundary. χ/Q is calculated using the well-established formula for Gaussian Dispersion, which is applicable when the effluent is released at such a rate that it does not perturb the existing pattern of turbulent eddies in the atmosphere. This is the expected case for small releases such as are evaluated in Section 7.0. χ/Q was calculated using the formula:

$$\frac{\chi}{Q} = \frac{1}{U_{10}\pi\Sigma_y\sigma_z} \qquad (9\text{-}1)$$

where: Σ_y = $M\sigma_y$, for distances of 800 meters or less and the value of M is determined from Figure 3 of NRC Regulatory Guide 1.145 (M=1 for all cases where the wind speed is 6 meters per second or more). For a wind speed of 2 m/s and atmospheric stability class F, M=4.

Then for a wind speed of 2 m/s:

$$\frac{\chi}{Q} = \frac{1}{U_{10}\pi M\sigma_y\sigma_z} \qquad (9\text{-}2)$$

where: σ_y = the lateral plume spread in meters (m)

σ_z = the vertical plume spread in meters (m)

U_{10} = the wind speed (meters/second) measured at a height of 10 meters.

The value of σ_y is determined from Figure 1 of NRC Regulatory Guide 1.145 and σ_z from Figure 2 of NRC Regulatory Guide 1.145. The values are a function of the distance, d, from the source and Pasquill's turbulence types. For Ward Center a distance of 100 meters (328 feet) and category F (Moderately Stable) meteorological conditions were utilized. Using d = 100 meters for category F meteorological conditions yielded σ_y = 4.0 meters and σ_z = 2.2 meters. Using these values of σ_z and σ_y and a wind speed, u, of 2 m/s, Equation 9-2 yields

$$\frac{\chi}{Q} = \frac{1}{(2)\pi(4)(4.0m)(2.2m)} = 4.52\times10^{-3}\ \sec/m^3$$